The strategy for accomplishing the mass balance project at ICPP is as follows:

1) Utilize existing DOE and Bechtel BWXT LLC protocols, procedures, and controls.
2) Obtain and utilize existing staff specialists and support personnel.
3) Establish a structured approach to meeting the project goals including the use of key assumptions.
4) Ensure effective communication of progress, issues, and problem resolution through regular meetings with project personnel.
5) Coordinate with other sites and share results.

2.0 SITE HISTORICAL OVERVIEW

2.1 The ICPP is located near the center of the 900 square mile INEEL which was formerly the National Reactor Testing Station (NRTS). The plant occupies approximately one square mile near the test reactors in an area that had formerly been used by the Navy for test firing large guns following relining of the barrels. The current facility/layout is shown in Figure 1.

2.2 Key Uranium Processing Facilities

The ICPP corner stone was laid in 1951. The Atomic Energy Commission (AEC) contractor during construction was the American Cyanamid Corporation. The construction contractor was the Blaw-Knox Company. The facility was designed by personnel at the Oak Ridge Laboratory Facility. In February of 1953 the first fuel (a slug from a Hanford production reactor) was charged to the dissolver. The dissolver product was purified using three cycles of methyl isobutyl ketone (hexone) extraction in packed columns. The acidic first cycle waste was stored in a cooled, 300,000-gallon, stainless steel tank located in a concrete vault. The acidic second and third cycle waste was stored in a second 300,000-gallon, stainless steel tank located in a separate concrete vault. The product from this processing campaign was sent to the Y-12 facility at Oak Ridge to determine whether the product met the acceptance criteria. It was subsequently accepted, and the plant began processing fuel. The plant processed fuel from that initial campaign in 1953 until 1992 when fuel reprocessing was discontinued by a secretarial edict from then DOE Secretary James Watson. A clean-out campaign was completed in 1996 and the product from that campaign,which only recovered uranium from solutions in storage in the plant, is still in storage at ICPP.

The historical development of the uranium recovery process is shown in Figure 2.

Figure 1
THE IDAHO CHEMICAL PROCESSING PLANT AS IT EXISTS TODAY

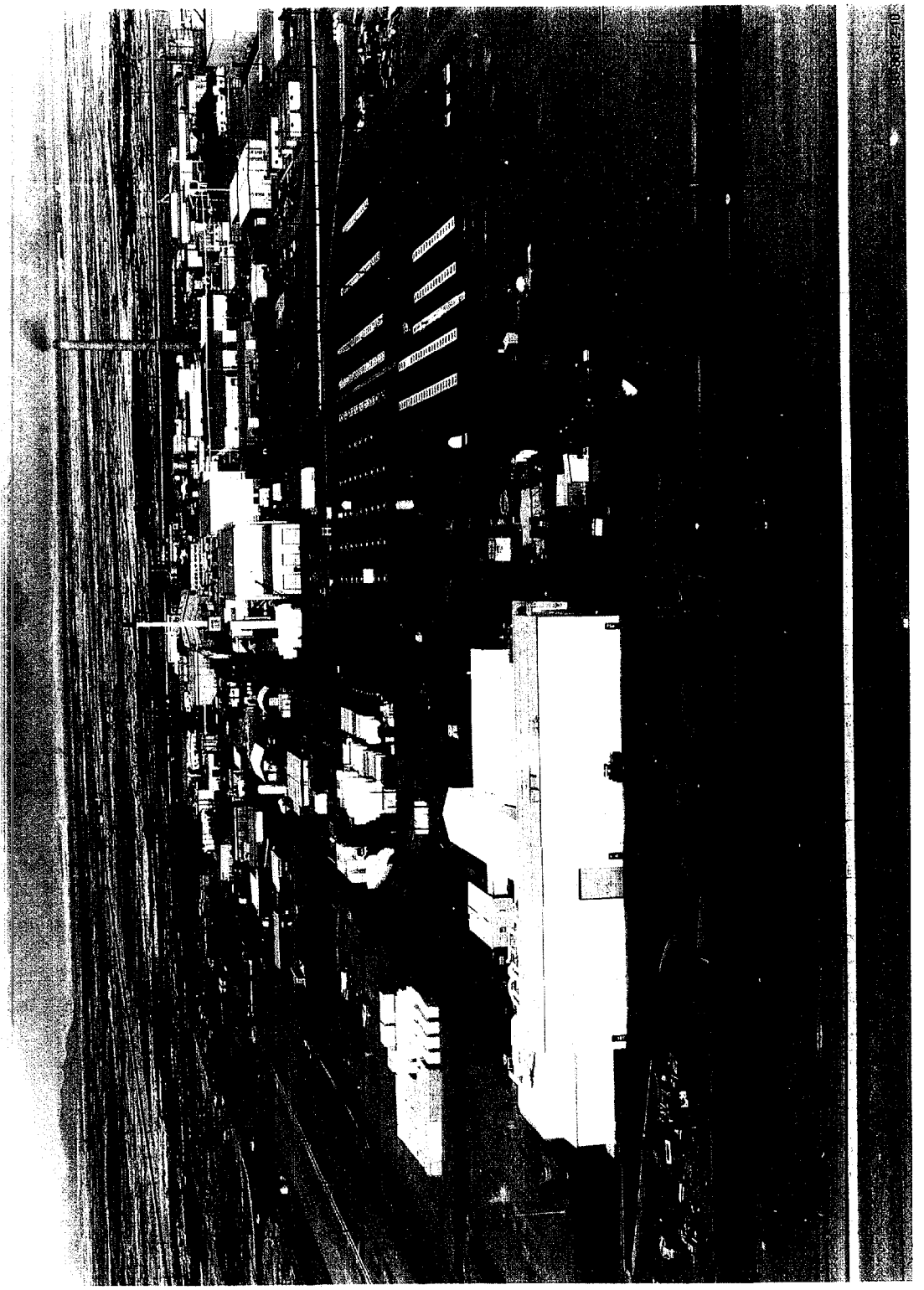
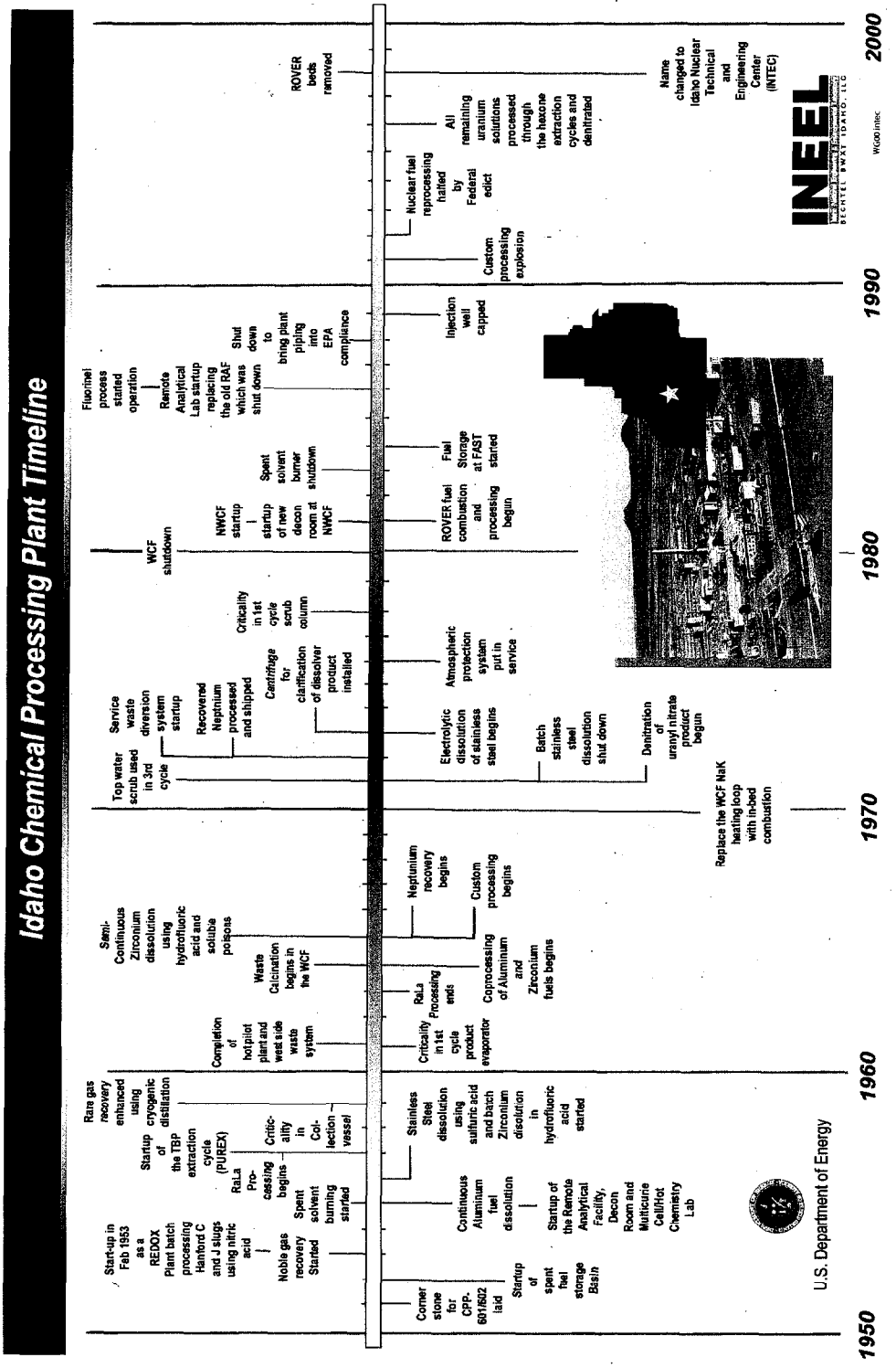


Figure 2
HISTORICAL TIME LINE OF IMPORTANT EVENTS AT
IDAHO CHEMICAL PROCESSING PLANT

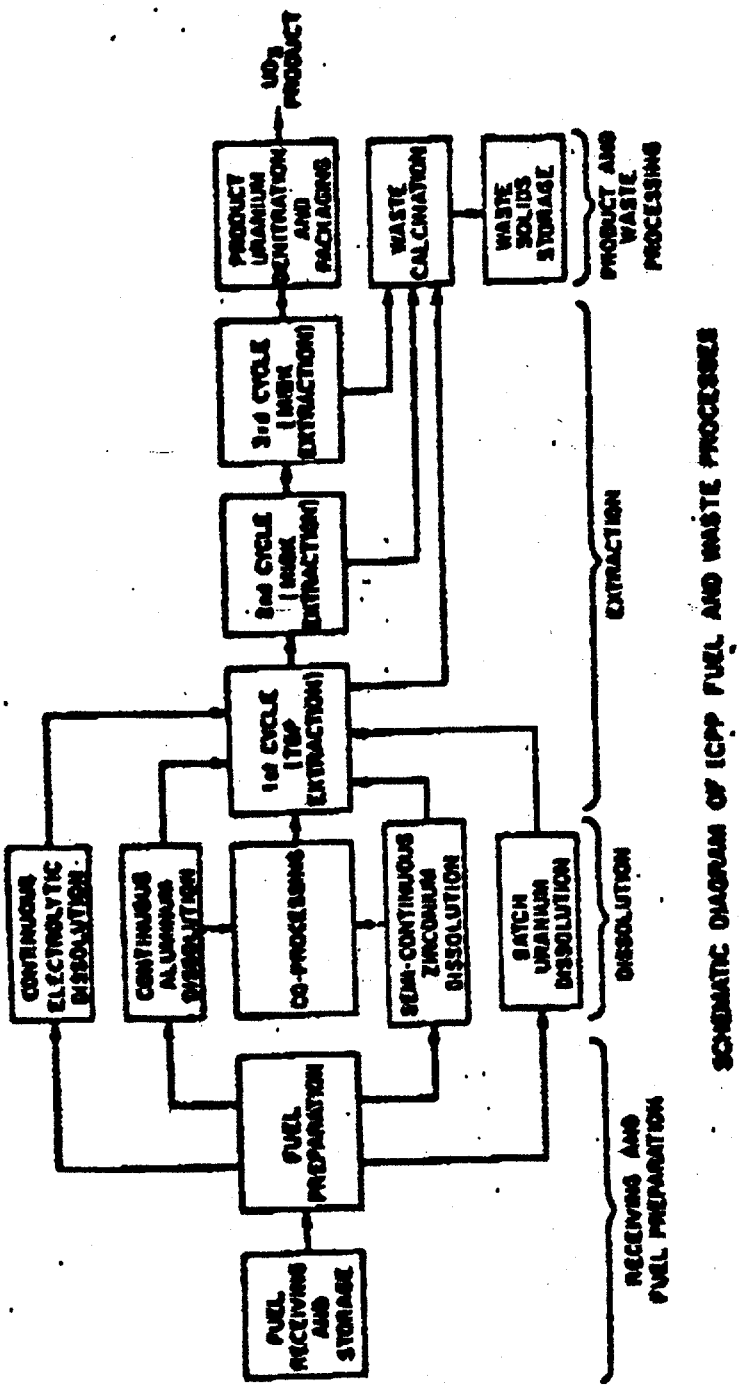


2.2.1 Idaho Chemical Processing Plant

2.2.1.1 Plant Description

The ICPP was originally built to process aluminum fuel from the Materials Test Reactor (MTR), unclad Experimental Breeder Reactor I (EBR-I) fuel, and Hanford neutron producing (NP) fuel using a methyl isobutyl ketone (hexone) extraction process. This process was used for the first seven processing campaigns. Dissolvers and extraction systems were all located in the CPP-601 processing building. The extraction system that was common to all dissolution processes at that time consisted of three cycles of methyl isobutyl ketone extraction using stainless steel, Raschig ring packed columns with a thermosyphon evaporator at the beginning of each cycle, and a product evaporator at the end of the third cycle. Typically, uranyl nitrate solution was fed to each extraction cycle at a concentration of approximately 250 grams of uranium per liter. The final product was shipped at a concentration in excess of 250 grams per liter. Bottling, sampling, and product storage were carried out in rooms in the basement of CPP-602. The 10-liter polyethylene bottles were weighed on a large, double-pan balance, then put into birdcages for shipment. The dissolution and extraction process for aluminum fuel was carried out in CPP-601 from 1953 until the plant was shut down in 1992. Product packaging operations were performed in CPP-602 for all processes. Appendix A contains flowsheets for all of the processes described in this section. A block diagram of the processes used at ICPP is shown in Figure 3.

Figure 3
FLOWSHEET OF PROCESSES USED AT ICPP



Processes for the dissolution of bare uranium slugs, declad EBR-I stainless fuel, aluminum clad fuel, batch dissolution of zirconium fuel, and the Radioactive Lanthanum (RaLa) process to recover radioactive barium from short cooled aluminum clad test reactor fuel were all started up during the first seven campaigns, process support modifications also took place. Analytical chemistry and process development laboratories went from standard 1950s style open bench-top laboratories with hoods to a Remote Analytical Facility (RAF) with shielded boxes utilizing castle manipulators and a development laboratory with RAF style boxes and a large process development cell with masterslave manipulators. Both of these modifications reduced exposure and risk of contamination. A large steel-lined room was also provided to decontaminate pieces of equipment used in the process facilities. These facilities were in CPP-627 until they were replaced with updated facilities in the 1980s. A process to remove the rare gases krypton and xenon from the dissolver off-gas using liquid nitrogen-cooled activated charcoal beds was also started up and operated. This process was located in CPP-604.

In 1955, the Continuous Processing Modification Project (CPM) was completed and a new high-capacity, first-cycle extraction system using tributyl phosphate dissolved in kerosene was placed into service. No preconcentration of the first cycle dissolver product was necessary, and the system could be operated concurrently with the fuel dissolvers. This system helped control criticality safety in the first cycle through the formation of stoichiometric compounds with the tributyl phosphate.

More modifications to the processes were made from 1957 to 1970. In 1958, the rare gas recovery plant was enhanced by replacing the carbon beds with a cryogenic distillation system. The recovery process for recovering radioactive barium was shut down in 1963. In 1964, the Waste Calcination Facility (CPP-633) was started hot to convert the high level wastes generated by the extraction columns and other radioactive liquid waste generating operations into a dry, granular waste form suitable for long-term storage. Custom processing in CPP-627 of small lots of odd fuel materials unsuitable for recovery anywhere else in the complex, semi-continuous zirconium dissolution in hydrofluoric acid containing a boric

acid neutron poison for criticality safety, and the recovery of neptunium from the second cycle waste for use as an irradiation target started up in 1965. In 1970, the sodium-potassium eutectic alloy heating loop in the Waste Calcination Facility was removed from the calciner vessel and replaced with an in-bed combustion system, which increased throughput and reduced nitrogen oxide and ruthenium emissions.

Two major innovations affected product and product shipments. The denitrator process in CPP-602 was started in 1971 with a fluidized bed thermal conversion process for converting uranyl nitrate to uranium trioxide. The entire process (denitration through product sampling and loadout) was enclosed in a glove box in the former uranyl nitrate bottling area of CPP-602 and operated until the process was shut down in 1996. Glove box operation minimized the potential for dust contamination to operating personnel.

A second innovation that also significantly affected the quality of the final product was put in service in 1971. This was a top water scrub that entered the top of the combination extraction/scrub column on the third extraction cycle (the second hexone extraction cycle). This scrub, whose original purpose was to reduce the amount of carryover of aluminum into the final product, also allowed the use of a second cycle for partitioning the higher actinides from the uranium since the iron from the ferrous ion reductant would not be carried over into the final product. This second partitioning cycle significantly reduced the amount of higher actinides in the product, as well as the carryover of fission products simply by removing entrained aqueous droplets being carried into the strip column by the organic product stream. In 1971, the batch, sulfuric acid, stainless steel dissolution headend was shut down.

In 1972, the neptunium that had been recovered from the second cycle partitioning step since 1965 was cleaned up using two cycles of hexone. The flow sheet used an acidic scrub rather than the normal acid deficient scrub to minimize losses of neptunium. Approximately 6.6 Kgs of neptunium was shipped to the Savannah River Site for use as targets in making Pu-238. The processing of neptunium was carried out in CPP-601. Bottling of the product was done in the multi-curie cell in CPP-627.

In 1973, the electrolytic dissolver for the dissolution of the stainless steel clad EBR-II fuel was put in service in CPP-640. In conjunction with the electrolytic dissolver, a centrifuge for the clarification of the electrolytic dissolver product was also put in service. Dissolution of the EBR-II fuel resulted in small grains of stainless steel that did not dissolve and a significant quantity of finely divided fissium solids being present in the dissolver product. The centrifuge was essential to successful operation of the extraction process for the EBR-II fuel. The product from this process was a low burnup (~ 2 atom %), lower-enriched (~ 50% enriched) UO_3 than was normally seen in the product. This product (~ 4.076 tonnes) was processed and packaged as a unique material and shipped directly to the Portsmouth Gaseous Diffusion Plant.

In 1981 the original Waste Calcination Facility (WCF) for processing high level waste was shut down for the last time. Decommissioning activities were completed and a concrete cap poured over the site in 1999. The New Waste Calcination Facility (NWCF) located in CPP-659, and a new decontamination room, built as a part of the NWCF to replace the original decontamination room in CPP-627, were started up in 1982. The new calciner featured a larger, fluidized-bed, calcination vessel for higher throughput and more remote maintenance capability for the remote replacement of failure prone equipment, which significantly reduced down time. It was shut down on May 26, 2000 pending permitting as an incinerator. There is currently approximately one million gallons of liquid waste left in storage at ICPP.

In 1983, the process for recovering uranium from the ROVER (Nuclear Rocket) fuel was started up. The ROVER fuel was a graphite rod with the uranium particles dispersed throughout the rod. The rods, which had been packaged in cardboard tubes, were burned in the primary burner. The ash from this burner was transferred to a secondary burner, where additional carbon was burned away prior to the ash being transferred to a leaching vessel. In the leaching vessels, the uranium was put into solution using a nitric/hydrofluoric acid mixture. It was extracted through the three cycles of extraction and then denitrated to UO_3. Part of the product was shipped to Los Alamos for criticality studies and the rest was sent to Y-12. The fuel had a very low burn up (~ 0.1%) and, thus, did not

have a significant buildup of either fission products or the actinides. This process operated for 14 months and was shut down. The fluidized bed burners have just recently been cleaned out.

In 1986, the Fluorinel Dissolution Process (FDP) was started up in CPP-666 to process zirconium-clad fuel. FDP had three large dissolvers that dissolved fuel in a mixture of hydrofluoric acid/aluminum nitrate, which had both boron and cadmium present as nuclear poisons. The Remote Analytical Laboratory (RAL) in CPP-684, was built to handle the sample load from the three FDP dissolvers, was started up in 1986. At the same time, the old Remote Analytical Facility (RAF) in CPP-627 was shut down. Replacing the RAF in total resulted in a significant reduction in radiation dose to the analytical personnel in the laboratory. In 1977, the radiation dose averaged approximately 500 mrem/person who worked with radioactive samples in the labs and the maximum was 1.2 rem on one individual. The first full year in the new lab that was concurrent with a processing campaign (1987) the average exposure was 30 mrem/person/year with the maximum about 300 mrem. However, in the ten-year period between 1977 and 1986 the average dose had slowly decreased as procedures, work practices, and equipment were changed. But, the largest decrease came with the new laboratory.

In 1988, the plant was temporarily shut down to bring the underground piping into compliance with EPA regulations. This entailed significant modifications throughout the processing facilities and the laboratories.

In 1991, the custom processing operation was shut down. In April 1992, an edict by then Secretary of Energy James Watkins halted all nuclear fuel reprocessing. The plant was, however, allowed to run the second and third cycle/denitration operation to completely remove all fissile material from the process tanks in 1996. That material and the material from the two Fluorinel campaigns is still stored in the CPP-651 vault.

In 1998, the ROVER beds were removed from the burners and uranium- containing materials from all of the other ROVER vessels was cleaned out. The ash is currently in dry storage at

the CPP-603 Irradiated Fuel Storage Facility (IFSF) awaiting disposition decisions. More than 100 Kgs of uranium is in this ash.

From 1953 until the recovery processes were finally shut down in 1996, all of the extraction processes, evaporative concentration processes, the product bottling, and the denitration process were operated in the CPP-601/CPP-602 buildings. Dissolution processes were operated in buildings CPP-601, CPP-640, and CPP-666. All of these processes were in heavily shielded cells in a totally remote operation. The dissolver system, the extraction systems, and the waste systems were all contact maintenance and depended upon extensive decontamination prior to cell entry. The liquid product bottling and the denitration product packaging operations were done in either a hood or a glove box, respectively. The flowsheets for all of the processes mentioned above, except for the waste processes, are shown in Appendix A.

2.2.1.2 Material Flowsheet

Spent fuel from reactors was originally received in CPP-603, which was a water filled storage basin. Other fuels were later received for dry storage in CPP-749 and eventually for dry storage in the IFSF an addition to CPP-603. In 1984, the water filled storage basin in the Fluorinel and Storage Facility (FAST), CPP-666, was started up and is currently storing spent fuel. The last fuel from the basins at CPP-603 was removed in May 2000 and the facility will soon be shut down and decommissioned.

After the decay of short-lived fission products including Iodine-131, the spent fuel from the storage basins at CPP-603 was transported to dissolvers in either CPP-602 or CPP-640. There, the fuel was dissolved in an acid specific to its particular cladding composition. Feed adjustments were made and the fuel was extracted initially in three hexone extraction cycles and later in a TBP/kerosene pulse column system followed by two cycles of hexone. The product from each extraction was concentrated by evaporation in a thermosyphon evaporator. The final product from the three extraction cycles was an aqueous solution of uranyl nitrate in nitric acid. After 1971, the uranyl nitrate solution was thermally decomposed in a

fluidized bed denitrator and shipped as a solid UO_3 granular product.

2.2.1.3 Feed Specifications

The feed to the ICPP was "as received" spent nuclear fuel. There were no acceptance criteria that determined whether the fuel was suitable for processing. In 1974, fuel receipt criteria were developed with a purpose of obtaining as much information on the fuel as possible to help understand the complexities associated with processing the fuel. Fuel could not be shipped until the receipt criteria response was provided, but responses to the questions would not prevent a fuel from being sent to ICPP.

2.2.1.4 Product Specifications

The early product specifications were informal and were subject to negotiation. A report by Egli, et. al. (Egli, 1985) suggested that a formal set of product specifications should be produced. This resulted in a letter (Foutch, 1985) from Y-12 to the managers of the plants at Savannah River and Idaho defining the specifications for the uranium product to be shipped to Y-12. These specifications defined the amount of alpha, beta, and gamma that could be in the product.

2.2.1.5 Operating History

The operating history of ICPP is detailed in Section 2.2.1.1.

2.2.1.6 Current Status

The process for recovering uranium from spent fuel is currently shut down. There is 1770 Kgs of uranium product in storage at ICPP. There are also several hundred Kgs of spent fuel stored in dry storage in CPP-749, CPP-603 IFSF and in wet storage in the CPP-666 fuel storage basin.

2.3 Activity Summaries

The primary concentrating process at ICPP was the extraction cycles that removed the fission products, activation products, and actinides from the uranium and then concentrated both the uranium by evaporation and the fission product waste streams either by evaporation and/or calcination.

A second product concentration process took the concentrated uranyl nitrate stream and denitrated it to uranium trioxide. Any contaminants in these streams that were not volatile were concentrated by the denitration process.

A third concentrating process was the ROVER burners. Graphite-based ROVER fuel was burned in fluidized bed burners resulting in an ash that contained uranium at a much higher concentration than was present in the fuel. A leaching process also may have resulted in a higher concentration.

Dissolution of the fuel in nitric acid could also result in a higher concentration per unit volume in the liquid phase than was present in the dry fuel state.

The above processes took place in remote equipment inside containment cells or boxes. Exposure to recycled uranium could occur after the product stream came out of the strip column in the last extraction cycle and was concentrated in an evaporator to 250 g/L or more.

An examination of the tailend processes that occur after the concentration of the product have identified processes where workers can be exposed to contaminants in the recycled uranium product. These areas and activities are described in Table I. An "occupational exposure potential value" is also given in the table. The potential for worker occupational exposure is expressed as high, medium, low, or none in the "Occupational Exposure Potential" column. This value is derived from the product of three parameters qualitatively assigned by the specific Site Team. Each Site Team reviewed activities at their site that might have exposed workers to increased levels of the constituents and answered the following questions:

1) How much (high, medium, low, or none) airborne dust is generated by the activity?
2) What is the radiological hazard (high, medium, low, or none) of the material generated by the activity?
3) What is the length of time (long, medium, or short) a worker would be exposed to the airborne materials?

Each variable was assigned a value for each question and the values were multiplied together to determine the Occupational Exposure Potential. Activities associated with long-term exposure to high levels of dust with high radiological activity received the highest score while short duration activities in clean areas received the lowest score.

The list in the following table represents those areas and activities that the site team believes presents the highest potential for worker occupational exposure.

Table I
ICPP Activity Chart

Building	Activity	Time Frame	Maximum Constituents Concentration	Occupational Exposure Potential
CPP-602	Bottling Liquid Product in a Hood	1953 - 1971	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-99	nil
CPP-602	Packaging Solid Product in a Glove Box	1971 - 1996	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-99	nil
CPP-627 CPP-602	Analysis of Liquid Product	1953 - 1996	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-99	nil
CPP-627 CPP-602 CPP-684 CPP-630	Analysis of Solid Product	1971 - 1996	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-97	M
CPP-602	Operating Denitrator	1971 - 1996	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-97	nil
CPP-602	Maintenance on Denitrator	1971 - 1996	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-97	M
CPP-602	Health Physics Surveillance on Denitrator	1971 - 1996	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-97	M
CPP-602	Health Physics Surveillance of Liquid Product Bottling	1953 - 1996	20% U-236 22 ppb Pu 1.6 ppm Np-237 1.8 ppb Tc-97	nil

2.3.1 Bottling Liquid Product

Liquid product, which was concentrated uranyl nitrate solution in aqueous dilute nitric acid, was bottled out in a hood in the basement of CPP-602. The hoods were tested to have a face velocity of 125 ft. per second, which was enough to prevent alpha recoil particles from escaping. The product, being in solution, also reduced the risk of airborne particulate contamination making this a nil risk operation.

2.3.2 Packaging Solid Product

The solid product packaging operation was carried out in a glove box in close proximity to the denitrator vessel. The product was accumulated in a vessel near the denitrator. When this vessel contained enough UO_3 to fill a shipping container, the UO_3 was transferred to a V-blender, which mixed and homogenized the UO_3 particles so that a representative sample could be obtained. The contents of the V-blender were then transferred to the shipping container. As the UO_3 flowed into the container, samples were taken for accountability analyses. When the transfer was complete, the shipping container was weighed, sealed, and bagged out of the glove box along with the samples. The shipping container was then put into the shipping box used to maintain spacing for criticality control. This was the package that was shipped to the other sites. This activity is also a nil risk operation.

2.3.3 Analysis of Liquid Product.

Because the solid product analyses required handling a particulate sample during the transfers and during weighing of the aliquot, it presented slightly more risk than the liquid analytical procedures, even though all of the operations with the final product were carried out in a hood. This operation was also classified as a nil risk.

2.3.4 Operating the Denitrator

The workers operating the denitrator were protected by the glove box that contained the denitrator process. Accordingly, even though the operators were in attendance during the operation, the risk was classified as nil risk.

2.3.5 Maintenance on the Denitrator

Some maintenance operations are carried out in the glove box, but others required disassembly of the process equipment. At those times, there could be more particulate contamination than in any other operations. Personnel were required to were personal protective equipment during those operations for protection. This operation is a medium risk operation.

2.3.6 Health Physics Surveillance During Denitrator Operation

Health Physics technicians monitor the radiation fields and air quality during denitrator operations. Their risk was essentially similar to those of themaintenance personnel and was classed as a medium risk.

2.3.7 Health Physics Monitoring During Liquid Product Bottling

Health Physics technicians faced lower risks during liquid product handling operations than that faced for solid product operations. These operations were classed as medium risk operations.

2.4 Work Force Exposure

All of the storage activities, processing activities, and waste processing activities were carried out in hot cells, so the radiation dose was carefully monitored and limited. Exposure to the product was limited through either handling in hoods when the product was bottled out as liquid or in a glove box for the uranium trioxide solid product.

The dose to the work force was due primarily to maintenance activities, processing activities, health physics activities, and analytical chemistry activities in the early years. Radiation doses were less than the allowed 5 rem/year. Subsequent to 1977 the practice was to limit dose to less than 3 rem/year to reduce the chance of challenging the 5 rem/year limit.

Analytical Laboratory dose in 1977 averaged 0.5 rem/year beta/gamma on personnel who were actively analyzing radioactive samples. The maximum dose that year was 1.2 rem/year. In 1987, the dose averaged 0.03 rem/year and the maximum was 0.1 rem/year. The reduction was the result of operating in a state-of-the-art remote analytical laboratory whose first full year of operation coincided with a major high-burnup spent fuel campaign. Because of the construction of a new state-of-the-art spent fuel storage and dissolution facility and a new state-of-the-art calciner, similar reductions in the radiation dose were experienced on the operations and maintenance staff.

The shift workers were the personnel at the highest risk for contamination or radiation dose. A paper by Reid, D., et al. (Reid, 1961) in the *Second Edition of the Reactor Handbook, Volume II* presents some insight into the staffing levels and radiation work practices at ICPP during the late 1950s. The shift worker staff consisted of 29 operations, 27 maintenance, 14 analytical, and 9 radiation control personnel in a staff of 265 personnel. By contrast an equivalent staff during the 1987 FDP campaign consisted of 104 operations, 36 maintenance, 28 analytical and 24 health physics personnel in a staff of 1800 personnel.

Radiation dose limits were pushed harder in the early days prior to the "as low as reasonably achievable" (ALARA) policy, as evidenced by the following remark from the paper by D. Reid, in the Reactor Handbook:

> It appears advantageous to utilize beta or gamma limits and to define such allowable limits over as long a period as practical. For example, a limit of an average of 5 rem/yr over a 10-yr period is much more useful than 100 mrem/week or 20 mrem/day. The problem of utilizing personnel to the best

> advantage under the limits is a serious one and takes planning, particularly in maintenance operations. For example, it is less advantageous to use a large number of men who will receive a very small exposure each than a smaller number of men receiving a larger individual exposure, since a significant fraction of the exposure will be received in setting set up to do the work before any useful maintenance is accomplished (page 648).

The quote seems to indicate that closely approaching maximum dose (5 rem/year) was not unusual and might have been expected for every worker. By the middle of the 1970s, radiation doses were lower, but the ALARA policy had come into being resulting in an awareness of radiation and a sensitivity to unnecessary radiation dose. In the 1970s, a major cleanup of the plant took place that changed the radiation zone designations around the plant. Areas that had been controlled were cleaned and managed as uncontrolled areas.

Another paper by D. R. Wenzel, et. al. (Wenzel, 1980) discusses radiation dose experience at ICPP from 1973 to 1978. This period was chosen because prior to 1973, ICPP was managed by a contractor whose contract with AEC covered most of the facilities at the INEEL. As the result, these contractors had the ability to move personnel from one area to another, in part to spread out contamination and in part to provide other opportunities for the personnel. However, this practice had the effect of making it very difficult to differentiate exposures that occurred at ICPP from those that occurred at the reactors or at the waste sites.

Wenzel's paper tracks production, maintenance, and health physics wherein analytical personnel were lumped in with the total plant personnel. During this time, the total plant dose varied from as little as 300 rem to as high as 680 rems Also, during this time, the monitored radiation worker population at the plant went from 600 to 1400 people. However, the change in production, maintenance, and health physics personnel was less than 10% from 230 to 290 people. During this same period, the average dose for health physics personnel was between 2.7 rem and 1.8 rem and was consistently about 1.2 rem through the 6 year period for both maintenance and production personnel.

During the period from 1973 to 1978, the total plant dose went from 375 rem to 640 rem. However, the demographics of the plant also changed. In 1975, a dedicated construction work force was used at ICPP. This increased the average dose of the construction workers at ICPP because of the smaller number of workers used on a larger number of radiation jobs. From 1976 to1978, the construction work force was approximately equal to the total maintenance, operations, and health physics workforce. In 1973, all other radiation workers received a total of 105 rem in 1973. By 1978, this had reduced to 69 rem for these

same "other" classifications. These classifications included management, technical, analytical, engineering and quality assurance.

The internal dosimetry program during this period consisted of whole body counts given annually to radiation workers. For personnel where internal contamination was suspected, formal dose assessments were made for cases where the calculated "fifty-year dose commitment" exceeded 10% of the radiation protection standard for any critical organ. Typically, the dose commitment levels were small fractions of the permitted limit of 15 rem per year, the total cumulative lung dose for any worker had not exceeded 8 rem/year, and the total for all workers has not exceeded 32 rem in a single year. The limiting internal contaminant had
been Ce-144.

In 1976, an administrative guideline of 3 rem per year was adopted that required special management approval to exceed this guideline. Administering this guideline required rapid processing of dosimeters with the up-to-date cumulative dose data managed on a computer. Access to the data by health physics personnel enabled them to control the dose from the field. Management was also alerted when any one individual began to approach the 3 rem guideline. Management of the work and the personnel was critical to maintaining cumulative exposure to less than the 3 rem.

In 1978, the dose equivalent for the total regular employee at ICPP indicated that no one had exceeded 4 rem that year, although there were 14 individuals between 3 and 4 rem. There were 67 people between 2 and 3 rem, 95 people between 1 and 2 rem, and 342 people who received a dose exposure greater than the minimum detectable amount up to 1 rem. There were also 430 people out of the total 948 total employees who received less than a detectable radiation dose. Radiation workers who received more than a detectable amount of radiation during 1978 were approximately 518 people.

A final note on radiation doses occurred in 1995 when the contractor at that time offered an early retirement incentive to employees 55 years of age or older. Approximately 350 people from the ICPP out of approximately 1800 total employees took advantage of the early retirement incentive. The effect on the cumulative radiation dose, however, was that slightly more than 50% of the cumulative radiation dose left with those retirees. An additional effect that this retirement offer had on the cumulative radiation dose was to significantly reduce the average dose per person by removing from the work force population, the "old timers" who had accumulated large doses at a time when the normal operating mode was to push the maximum annual dose limit.

An assessment of the relative risk to an individual handling ICPP product can be made using the data from the ORIGEN2 calculations for the three typical fuels processed at ICPP. The radionuclide distribution data was then entered into the RSAC-5 computer program to evaluate the relative amount of internal dose from each of the radionuclides.

An assumption was made that the transuranic alpha in the final product was 5000 disintegrations/minute (dpm) per gram of total uranium. A further assumption was made that the isotopic distribution of uranium did not change from the ORIGEN2 calculated values as the uranium was processed through the ICPP extraction systems. Still further, an assumption was made that the isotopic distribution of the plutonium did not change while processing and that the ratio of both neptunium-237 and technetium-99 to plutonium is the same as it is in the dissolver product.

Other assumptions were made to make the model fit the situation since the model the computer code uses is an airborne inhalation model. A rate of 3.33×10^{-4} cubic meters/second (m^3/s) was assumed for the breathing rate for an individual and an internal dose was assumed to occur over a 50 year time period. A particle size of 1.0 microns activity median acrodynamic diameter (AMAD) was assumed. The lung clearance class for this calculation is shown in Table II.

Table II
Lung Clearance Classes Used to Determine the Relative Hazard from Various Isotopes

Element	U	Np	Pu	Th	Am	Pa	Ra	Pb	Tc
Lung Clearance Class	Y	W	Y	Y	W	Y	W	D	W

Using these assumptions, the program calculated the committed effective dose equivalent (CEDE) for each radionuclide and its percent contribution to the total inhalation. This data is shown in Table III.

As can be seen from the Table III, the risk from inhalation is due primarily to the uranium isotopes at 5000 dpm transuranic alpha per gram of uranium. The plutonium isotopes have a risk on the order of the 10^{-3} % while the sum of the uranium isotopes have in excess of 99.9 % of the risk. Both neptunium-237 and technetium-99 are on the order of less than 10^{-17} % of the dose.

Because, this analysis was done using the maximum transuranic (TRU) activity allowed by the alpha specification (5000 dpm TRU/gram U), the actual percent of the dose from the actinides, Pu and Np will be actually less than indicated in Table III. For the product from aluminum and stainless steel processing, U-234 is the most limiting radionuclide. U-235, however, is the limiting radionuclide from the zirconium process. The potential dose from plutonium is more than three orders of magnitude less than from the dose from uranium. The dose from neptunium and technetium is insignificant compared to that from uranium.

The plutonium isotope that contributes the highest potential dose from inhalation of uranium product is Pu-238 for the zirconium and aluminum fuel processing and Pu-239 for stainless steel processing. However, the potential Pu-239 dose from the product of stainless steel processing is less than 0.02% of the dose from uranium.

Table III
Comparative Risk and Effective Dose Equivalent for Isotopes in the Product from Processing at ICPP.

Isotope	EDE (rem) Aluminum	Percent	EDE (rem) Zirconium	Percent	EDE Stainless Steel	Percent
U-232	1.72 E-7	3.82 E-1	6.92 E-7	8.90 E+0	1.59 E-8	9.29 E-2
U-233	4.18 E-10	9.29 E-4	1.29 E-10	1.66 E-3	1.19 E-10	6.98 E-4
U-234	4.03 E-5	8.96 E+1	3.28 E-7	4.22 E+0	1.64 E-5	9.59 E+1
U-235	7.80 E-7	1.74 E+0	7.80 E-7	1.00 E+1	5.05 E-7	2.95 E+1
U-236	3.72 E-6	8.27 E+0	5.97 E-6	7.68 E+1	1.01 E-7	5.90 E-1
U-238	1.24 E-8	2.76 E-2	2.57 E-9	3.31 E-2	7.34 E-8	4.29 E-1
Pu-238	6.10 E-10	1.36 E-3	2.63 E-9	3.38 E-2	6.53 E-11	3.81 E-4
Pu-239	9.59 E-12	2.13 E-5	1.53 E-12	1.96 E-5	2.91 E-9	1.70 E-2
Pu-240	5.75 E-12	1.28 E-5	1.20 E-12	1.55 E-5	2.16 E-11	1.26 E-4
Pu-241	3.73 E-11	8.29 E-5	2.08 E-12	2.68 E-5	2.20 E-13	1.29 E-6
Pu-242	1.17 E-14	2.60 E-8	6.46 E-16	8.54 E-9	5.22 E-19	3.05 E-12
Np-237	< E-22	< E-17	< E-22	< E-15	< E-23	< E-17
Tc-99	< E-22	< E-17	< E-22	< E-15	< E-23	< E-17
Total	4.5 E-5 rem	100.0 %	7.77 E-6 rem	99.9 %	1.71 E-5 rem	100.0 %

In the 1980s, a fecal sampling program was added to routine urine sampling that had been in place since the 1950s. The early fecal sampling that started in 1980 identified internal contamination in analytical laboratory personnel that was traced to a bad hood and hot cell ventilation system. Occasional internal contamination incidents have occurred through the years with radiation doses at levels slightly above background.

2.5 Environmental Releases

The *INEEL Historical Dose Evaluation report, Volume I,* (INEEL, 1991) attempted to determine the off-site dose that resulted from activities at the site. The site has released radionuclides through injection wells at the facilities since each individual facility started up. Radionuclides were never discharged in the surface waters such as the Big Lost River or Little Lost River. The practice of injecting waste water deep under-ground was stopped in 1984 with the closure and sealing of the ICPP injection well.

Radionuclide migration has been tracked through sampling the water in wells drilled into the aquifer all over the site. Two radionuclides are of particular interest, tritium, because it is a component of the water molecule and chlorine-36, because of its high solubility as the chloride ion and its long half-life (3.0×10^5 years). Chlorine-36 has been detected at the site boundary, but at levels that are one-millionth of the amount permitted by the EPA in community drinking water. Tritium has also been detected at wells at the site boundary, but has not been found in any off-site wells. Neither of these radionuclides has contributed any significant dose to any member of the public as the result of activities at the site by this route. In addition to tritium and chlorine-36, other radioactive elements such as plutonium, cesium, and strontium, were also considered but were found to absorb on the soils.

Some biotic pathways also exist, the most important being through big game animals that ingest water or plants contaminated with radionuclides and then migrate off site. Through a literature search on this pathway the dose reconstruction group concluded that this was a highly unlikely source of radiation exposure and could result in a dose as high as 10 mrem/hr.

In their assessment, the airborne pathway is the principal pathway for release of radionuclides to the public. Releases from the site were broken into two classes: operational releases and episodic releases. Operational releases are continuous releases that extend over the length of operating periods while episodic releases are the result of experiments, tests, or accidents and are typically short in duration and treated as distinct events.

Annual site releases varied from less than 10,000 Ci to as high as 1.5 million Ci released in 1961. Most of the activity was short lived consisting of noble gases and their particulate daughter products. This covered the forty year time period

from 1951 through 1990. Operational site releases peaked from 1957 to 1959 and have declined by approximately two orders of magnitude through 1989. The episodic dose contribution was less than 1%, except between 1955 through 1961. During the entire forty year period that the dose reconstruction report covers, there have only been two ICPP events that contributed more than 0.1 mrem to the annual dose. These two events were the criticality accident that took place on October 16, 1959 and the fuel element cutting facility (FECF) filter break that occurred on October 29-30, 1958.

The effective dose equivalent (EDE) from the FECF filter break for an adult, child, and infant was 0.11, 0.12, and 0.12 mrem, respectively. The maximum organ dose (to the skin) was 1.4 mrem irrespective of age. The EDE for the 1959 criticality event for an adult, child, and infant was 1.1, 1.2 and 1.5 mrem, respectively; and the thyroid dose, which was the maximum organ dose, for the adult, child and infant was 6, 9, and 22 mrem. These dose estimates were based on the assumption that the people were living on the boundary of the site full time.

The period when the operational dose from the ICPP was contributing a significant amount to the off-site dose was during the early years of the RaLa process - specifically between 1957 and 1959. During those years the EDE was predominantly due to I-131, which was released during RaLa processing of fresh fuel to recover the short-lived barium-140. By 1959, the off-gas tank for delaying the release off-gas from the dissolution until the I-131 decayed, was in place and had reduced I-131 emissions that year by a factor of two.

In spite of the various episodic releases and the operational releases, there has not been any year in the history of the INEEL site that the radiation doses exceeded the applicable public dose standards in place during that year. During the late 1950s, the EDE may have been as high as 9% of the whole body dose standard and as high as 90% of the organ dose standard. During the more recent years, when more restrictive standards have been in place, the off-site dose to the maximally exposed person has been less than 1% of the whole body standard and less than 3% of the organ dose standard. These doses are insignificant when compared to the natural background doses for a person living on the Snake River Plain. The natural background is about 350 mrem/yr due to terrestrial, cosmic, naturally occurring radionuclides and radon sources. The maximum EDE occurring in 1956 from airborne releases at the INEEL was 17% of the natural background level. Since the 1970s, the doses have been very small, even compared to the variability of the natural background from year to year and from location to location in Eastern Idaho.